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U.S. Nuclear Regulatory Commission
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Donald C. Cook Nuclear Plant Units 1 and 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION ON
LICENSE AMENDMENT REQUEST FOR MODIFICATIONS TO
AUXILIARY FEEDWATER PUMP ROOMS
(TAC NOS. MA8183 AND MA8184)

References:
1) I&M to NRC letter C0200-04, "License Amendment Request – Modifications to Auxiliary Feedwater Pump Room Cooling," dated February 18, 2000.

2) NRC to I&M letter, "Donald C. Cook – Summary of March 6, 2000, Public Meeting and Resulting Request for Additional Information Regarding an Unreviewed Safety Question Associated with Modifications to the Auxiliary Feedwater Pump Rooms (TAC Nos. MA8183 and MA8184)," dated March 14, 2000.

In Reference 1, Indiana Michigan Power Company (I&M), the Licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, proposed to amend Facility Operating Licenses DPR-58 and DPR-74. I&M proposed to modify the auxiliary feedwater (AFW) pump rooms to protect the equipment in the rooms from the environmental effects of a postulated high-energy line break (HELB). This would be accomplished by sealing the AFW pump rooms to ensure that the rooms do not communicate with the turbine buildings or each other. Sealing these rooms results in the need to modify the ventilation systems for the AFW pump rooms. The proposed AFW pump room modifications involve an unreviewed safety question in accordance with 10 CFR 50.59 since the probabilities of malfunction of the new cooling systems for the AFW pump rooms are higher than those for the current ventilation equipment. Therefore, NRC staff review and approval are required.

A001

On March 6, 2000, representatives of I&M met with the staff at NRC Headquarters in Rockville, Maryland. I&M discussed supplemental information in support of the license amendment request. In Reference 2, the staff provided a summary of this meeting and requested additional information. I&M is providing the requested information in the attachment to this letter.

I&M has evaluated the attached information and concludes that the evaluation of significant hazards considerations contained in Attachment 2 to Reference 1 is not affected. There are no new commitments made in this submittal.

Should you have any questions, please contact me at (616) 466-2698 or Mr. Walter T. MacRae at (616) 697-5633.

Sincerely,

Robert C Godley

R. C. Godley
Director of Regulatory Affairs

Attachment

\dms

c: J. E. Dyer
MDEQ - DW & RPD, w/o attachments
NRC Resident Inspector
R. Whale, w/o attachments

ATTACHMENT TO C0300-16

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Indiana Michigan Power Company (I&M), the Licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, provides the following response to the NRC letter, "Donald C. Cook – Summary of March 6, 2000, Public Meeting and Resulting Request for Additional Information Regarding an Unreviewed Safety Question Associated with Modifications to the Auxiliary Feedwater Pump Rooms (TAC Nos. MA8183 and MA8184)," dated March 14, 2000.

NRC Question 1

"Provide details on how the modifications to the AFW [auxiliary feedwater] pump rooms have been installed in accordance with the [CNP] design and licensing basis and in compliance with applicable rules and regulations. The information should be provided in a short bullet format as agreed to in the meeting."

I&M Response to Question 1

The changes related to the modification are being made such that full compliance will be maintained with applicable aspects of the CNP design and licensing bases and in compliance with applicable rules and regulations. This is controlled by plant procedures used in the development of design changes and associated 10 CFR 50.59 safety evaluations. The specific aspects of the design and licensing bases adhered to during the design of the modification are outlined below.

Attribute	Licensing and Design Basis (As Applicable)
Missile Generation	• Updated Final Safety Analysis Report (UFSAR) 1.4.1.5, Missile Protection • UFSAR 10.5.2.3, Design Evaluation • Proposed Atomic Energy Commission (AEC) General Design Criteria (GDC) - 1967, GDC 2 and 40 (Final Safety Analysis Report (FSAR) Appendix H)
HELB Interaction	• UFSAR 14.4.10.4, Ventilation Protection from Turbine Building Environment • FSAR Appendix O, "Postulated Pipe Failure Analysis Outside of Containment"
Non-Essential Equipment/Components	• Seismic "2 over 1" is maintained
Single Failure Criteria and Redundancy	• UFSAR 10.5.2.3, Design Evaluation - AFW • Institute of Electrical and Electronic Engineers (IEEE)-279, Draft 1968 - IEEE Criteria for Protection Systems in Nuclear Power Generating Stations • IEEE-308, 1970 - IEEE Standard for Class 1E Electrical Power Systems for Nuclear Power Generating Stations • Proposed AEC GDC - 1967, GDC 19, 20, 21, 39, and 41 (FSAR Appendix H) • UFSAR 1.4.4, Reliability and Testability of Protective Systems
Electrical Diversity and Power Sources	• IEEE-279, Draft 1968 - IEEE Criteria for Protection Systems in Nuclear Power Generating Stations • IEEE-308, 1970 - IEEE Standard for Class 1E Electrical Power Systems for Nuclear Power Generating Stations • UFSAR 10.5.2, Auxiliary Feedwater System • Proposed AEC GDC - 1967, GDC 39 (FSAR Appendix H)

Attribute	Licensing and Design Basis (As Applicable)
Piping/Vessel Pressure Boundary	• American National Standards Institute (ANSI) B31.1.0, Power Piping, 1967 • American Institute of Steel Construction (AISC) Manual of Steel Construction • Manufacturers Standardization Society of the Pipe, Valve and Fitting Industry, MSS-SP-58
Internal Flooding	• UFSAR 14.4.2.6.2, Flooding • ANSI B31.1.0, Power Piping, 1967
System Leakage	• FSAR Questions and Answers (Q&A), Q 4.22
Testing	• UFSAR 13.3.4, Post Startup Surveillance and Testing Requirements • I&M response to Generic Letter 89-13
Instrument Setpoint Sensitivity Control Philosophy	• FSAR Q&A, Q 7.18 • Proposed AEC GDC - 1967, GDC 12 (FSAR Appendix H) • I&M Commitment: AEP:NRC:1260:G3
Auto-Actuation of AFW System	• UFSAR 10.5.2, Auxiliary Feedwater System
Manual Actuation of AFW System	• UFSAR 10.5.2, Auxiliary Feedwater System
Auto Termination of AFW	• UFSAR 10.5.2, Auxiliary Feedwater System
AFW Capacity	• UFSAR 10.5.2, Auxiliary Feedwater System
AFW Technical Specifications	• Not required in accordance with 10 CFR 50.36
NUREG-0611 and NUREG-0635	• N/A
NUREG-0737	• N/A
Hot Shutdown for 4 hours	• UFSAR 3.3.1.5, Shutdown Margins • Proposed AEC GDC - 1967, GDC 26 (FSAR Appendix H)
Appendix R/Safe Shutdown Capability	• 10 CFR 50, Appendix R, Section III.G • IEEE-279, Draft 1968 - IEEE Criteria for Protection Systems in Nuclear Power Generating Stations • Underwriters Laboratories (UL) Ratings for barriers • Branch Technical Position (BTP) Auxiliary Power Conversion System Branch (APCSB) 9.5-1, Appendix A
Station Blackout/Loss of Off-Site Power	• UFSAR 8.7, Station Blackout • UFSAR 8.7.2.4, Effects of Loss of Ventilation • 10 CFR 50.63, Loss of All AC Power • Nuclear Management and Resources Council (NUMARC) 87-00 and Regulatory Guide 1.155 used to evaluate CNP against Station Blackout Rule
Seismic Qualification	• UFSAR 1.1.4, Seismology • ANSI N45.2.2 - "Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants During the construction Phase" • IEEE-323, 1974 - IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations • IEEE-344, 1975 - IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
SQUG	• I&M responses to GL 87-02, Supplement 1
Environmental Qualification	• UFSAR 14.4, Environmental Qualification • IEEE-323, 1974 - IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations for 10 CFR 50.49 • IEEE-323, 1971 - IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations for I&E Bulletin 79-01B (DOR Guidelines)
Maintenance/Preventive Maintenance	• In accordance with the Vendor recommendations and as incorporated into the CNP Preventive Maintenance Program
Inservice Inspection/Inservice Testing (ISI/IST) Program	• UFSAR 10.5.2.4, Tests and Inspections • American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI • Beginning the 3rd 10-year ISI interval, pump and valve tests are to be conducted in accordance with ASME OM Standards and NUREG-1482.
Personal Safety	• Occupational Safety and Health Administration (OSHA) 29 CFR 1910.146
Abandoned Equipment	• 10 CFR 50.59, Changes, Tests, and Experiments

Attribute	Licensing and Design Basis (As Applicable)
ALARA	• UFSAR 11.4.2, Radiation Control • 10 CFR 20, Standards for Protection Against Radiation
Human Factors	• Engineering Change Package (ECP) 1-2-02-01, "Control Room Human Engineering Criteria Report" • PMI-2115, Revision 1, "Plant Labeling"
Fire Protection	• UFSAR 1.4.1.3, Fire Protection • PMP-4030.ATR.001, Revision 4, "Administrative Technical Requirements" • 12-EHP-2270-FIRE.002, Revision 0, "Maintenance and Control of Fire Protection and Appendix R Documents" • Specification DCC-FP-101-QCN, Revision 14, "Fire Barrier Penetration Seals" • Specification ES-FIRE-0602-QCF, Revision 0, "Fire Rated and Non Rated Seismic/Expansion Gap Seals" • Proposed AEC GDC - 1967, GDC 3 (FSAR Appendix H)
Training	• 10 CFR 55.59, Licensed Operator Requalification
Emergency Lighting	• UFSAR 8.3.6, Lighting System • 10 CFR 50, Appendix R, Section III.J
PRA	• I&M Response to GL 88-20, Individual Plant Examinations
Emergency Operating Procedures	• Westinghouse Emergency Response Guidelines (ERGs), Revision 1C
Vendor Equipment Recommendation	• Incorporated into the CNP Preventive Maintenance Program
Maintenance Rule	• 10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants
Electrical Loads	• UFSAR 8.4, Electrical Systems/Emergency Power System • UFSAR 8.5, Electrical Systems Design Evaluation
EDG Loading Sequence	• UFSAR 8.4, Electrical Systems/Emergency Power System
Circuit Breakers/Protection Relays	• IEEE-323, 1974 - IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations • IEEE-344, 1975 - IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations • Regulatory Guide 1.100, August 1974 • National Electrical Manufacturers Association (NEMA) Standard ICS 2.3, 1974
Electrical Configuration	• UFSAR 8.1.1, Electrical Systems Design Criteria • UFSAR 8.1.2, Electrical Systems Functional Criteria • IEEE-279, Draft 1968 - IEEE Criteria for Protection Systems in Nuclear Power Generating Stations • IEEE 308, 1970 - IEEE Standard for Class 1E Electrical Power Systems for Nuclear Power Generating Stations • Proposed AEC GDC - 1967, GDC 39 (FSAR Appendix H)
Effects of Power Interruptions	• (See Single Failure Criteria and Redundancy)
Plant Security Impact	• Regulatory Guide 1.17 • Regulatory Guide 5.66 • ANSI/ANS-3.3-1982
Fluid System Characteristics	• UFSAR 10.5.2, Auxiliary Feedwater System • CNP Flow Accelerated Corrosion Program
Heat Exchanger Performance	• I&M response to Generic Letter 89-13
HVAC Systems	• American National Standards Institute/American Welding Society (ANSI/AWS) D1.3, 1998 - Sheet Metal Structural Welding Code • ANSI/AWS D1.3, 1992 - Steel Structural Welding Code • NEMA MG 1, 1998 - National Electrical Manufacturers Associations Standards Publication • American National Standards Institute/American Society of Heating and Air Conditioning Engineers (ANSI/ASHRAE) 15, 1994 - Safety Code for Mechanical Refrigeration • ASME NQA-1, 1989 - Quality Assurance Program Requirements for Nuclear Facilities

NRC Question 2

"Provide justification showing that with an increase in the probability of a malfunction associated with the AFW pump room modifications, that the AFW system and any other systems affected by the modifications will continue to perform their intended safety function as described in the UFSAR [Updated Final Safety Analysis Report]."

I&M Response to Question 2

The dominant impact of the proposed modification is a minimal increase in the probability of failure of the AFW pump room cooling function. However, operation of the AFW pumps is not adversely affected by this minimal increase. It has been demonstrated that the AFW pumps can perform their intended safety function for at least four hours for the turbine-driven AFW pump (TDAFP) and one hour for the motor-driven AFW pump (MDAFP) without room cooling. This allows sufficient time for operator actions to recover room cooling by restoring the room coolers to operation or by opening the AFW pump room doors. There is no direct impact of the proposed modification on the capability of the AFW pumps to automatically actuate and deliver required AFW flow for any of the accidents analyzed in the UFSAR. In addition, the proposed modification complies with the AFW system single failure criteria requirements. Therefore, the AFW system remains capable of performing its intended safety function as described in the CNP UFSAR.

The second possible impact of the proposed modification is on the operation of the essential service water (ESW) system. The proposed room coolers include supply and return lines from the ESW system, with manual throttle valves in the supply lines for performing ESW system flow balancing. An evaluation of the hydraulic response of the ESW system has been performed, conservatively assuming that the new supply line throttle valves to the room coolers are fully open. This evaluation demonstrates that the ESW system is capable of supplying required ESW flows to all components cooled by ESW, including the new room coolers. In addition, the proposed modification does not add any active components to the ESW system, and does not create any new failure modes for the ESW system. The new supply and return lines have been designed and will be installed in compliance with the CNP design and licensing basis for the ESW system. Therefore, the ESW system remains capable of performing its intended safety function as described in the CNP UFSAR.

The minimal increase in the probability of failure of the AFW pump room cooling function, and possible impacts on AFW and ESW operation, are more than offset by several mitigating factors. First, the proposed modification will protect the AFW pumps from the effects of postulated high-energy line break (HELB) events in the turbine building. Second, the proposed modification will protect each MDAFP from the effects of a postulated HELB in the TDAFP room. Finally, the proposed modification includes a requirement to enhance procedures for responding to a loss of AFW pump room cooling and for a loss of ESW header or system. The effect of these mitigating factors have been demonstrated to result in an overall reduction in the risk of core damage as further discussed in response to Question 3F below.

NRC Question 3A

"Provide the following information associated with the risk assessment:

A. Provide a copy of the risk assessment performed to support the proposed modifications. Describe the general methodology used for the risk assessment. The description should also include how the shortcomings and limitations of this methodology, as compared to an updated state-of-the-art PRA [probabilistic risk assessment] model, do not significantly underestimate the risk impact."

I&M Response to Question 3A

The risk assessments to support the proposed modifications have been performed following several different approaches. The first two approaches involved the use of simplified Loss of ESW and outside containment HELB event trees to provide a relative measure of the risk impact of the proposed modification and enhanced operating procedures. The simplified Loss of ESW event tree was the basis for the risk assessment results presented in I&M to NRC letter C0200-04, "License Amendment Request – Modifications to Auxiliary Feedwater Pump Room Cooling," dated February 18, 2000, and was discussed in the March 6, 2000, meeting between I&M and NRC staff. In addition, a third risk assessment involving the use of the CNP full-scope PRA model has been performed in response to this question.

The limitations of using the simplified Loss of ESW and outside containment HELB event trees include the possibility of overlooking a dependence between top events and underestimating core damage frequency (CDF) for some sequences. In addition, the focus of the simplified risk assessment is limited to a small subset of initiating events. However, the overall change in risk as determined by use of the simplified event trees versus the use of the CNP full-scope PRA model is not significantly different. This demonstrates that the limitations involved in the simplified event trees do not significantly impact the estimated change in CDF resulting from the proposed modification.

NRC Question 3B

"Provide the following information associated with the risk assessment:

B. Provide all significant assumptions and their justifications made in the risk assessment methodology. For example, the operability of the AFW pumps without room cooling is *important, and so is the potential operator recovery of the AFW system upon loss of Essential Service Water (ESW) by opening the AFW pump room doors."*

I&M Response to Question 3B

The common cause failure (CCF) value for the entire ESW system was reviewed and determined to be inappropriately conservative. A scoping estimate based on an appropriate grouping of the components in the system results in an ESW CCF on the order of 1 x 10^{-10} per reactor year (i.e., a reduction of five orders of magnitude from the existing CCF value). The only credible CCF mode for ESW with a higher likelihood was judged to be due to an environmental event. Based on the clean water supply for the ESW system and the lack of any historical record of environmental events affecting ESW at CNP, the ESW CCF probability was assumed to be 1 x 10^{-7} per reactor year.

The HELB initiating event frequency (IEF) value used is 1 x 10^{-2} per reactor year, as described in NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 - 1995." This value is believed to be conservative for CNP because of the flow accelerated corrosion monitoring program. The HELB events that could cause failure of the AFW pumps were determined based on engineering judgement and walkdowns of HELB piping.

Human error probability (HEP) values were determined for the operator action to open MDAFP room doors in response to a loss of room cooling. The HEP for opening the TDAFP room doors following a loss of room cooling was judged to be negligible compared to the TDAFP random failure probability (~0.1). An HEP value was determined for the enhanced loss of ESW procedure that was drafted during the risk assessments.

NRC Question 3C

"Provide the following information associated with the risk assessment:

C. *Provide a short description of the information on plant design features and operating experience, i.e., reliability, unavailability, and/or events, associated with the ESW system. How does the information compare to the loss of ESW frequency used in your analysis?"*

I&M Response to Question 3C

The ESW system is shared between both units, and consists of four ESW pumps and duplex strainers, and associated piping and valves. System piping is arranged in two independent headers. The two headers are arranged such that a rupture in either header will not jeopardize the safety functions of the system. Each header is served by one ESW pump from each of the two units. One header is sufficient to supply all service water requirements for unit operation, shutdown, refueling, or post accident operation, including a loss-of-coolant accident (LOCA) on one unit and a simultaneous hot shutdown in the other. All pumps receive a start signal in the event of an accident. Each ESW pump, associated strainer, and associated motor-operated valves have the same electrical power supply from either normal or emergency diesel generator power sources. In addition to electrical power, the ESW system also depends on the control air system to operate several ESW system valves.

The ESW system is a highly diverse and redundant system, with cross-connect capability between the units. The raw water source, Lake Michigan, is a very clean source of water. There is only one recorded event in which ESW header pressure decreased for unknown reasons such that the standby ESW pump automatically started on low header pressure. These characteristics of the CNP ESW system are reflected in the Loss of ESW IEF that was determined in the CNP full-scope PRA model. Due to the highly redundant design of the system, several random failures of the various components are required to result in a loss of ESW event. In addition, common cause failures specific to each header were included in the fault tree model. Therefore, the CNP full-scope PRA model is conservative with respect to the historical and predicted reliability and unavailability of the ESW system. The final result of this analysis is a loss of ESW IEF of 2.79×10^{-5} per reactor year.

NRC Question 3D

"Provide the following information associated with the risk assessment:

D. *The operator action to recover AFW pump room cooling in the event of a loss of ESW is risk significant. Do the proposed design modifications include appropriate procedural changes and operator training for operators? Describe the loss of ESW Emergency Operating Procedure enhancement."*

I&M Response to Question 3D

The proposed plant modification includes a requirement to enhance the alarm response procedure (ARP) for responding to a high temperature alarm in the AFW pump rooms. In addition, the proposed plant modification includes a requirement to enhance the abnormal operating procedure (AOP) for responding to a loss of ESW header or system. The enhancements to the existing AOP expand applicability to include loss of ESW flow for any reason, including a rupture in the ESW system. These procedure enhancements, including operator training, are required to be implemented prior to Mode 3 entry for each unit.

NRC Question 3E

"Provide the following information associated with the risk assessment:

E. Explain the source of the risk increase by describing the dominant sequences affected by the proposed modifications. The change in risk should also include the contribution from the full spectrum of initiating events including external events, not just from the loss of ESW initiating event."

I&M Response to Question 3E

The simplified Loss of ESW event tree estimates that the risk increase for the Loss of ESW event tree due to the proposed modification is approximately 2×10^{-6} per reactor year. This risk assessment compares the CDF for the base case Loss of ESW event tree to a case that assumes the TDAFP could survive without room cooling. The CNP full-scope PRA model Loss of ESW event tree shows a risk increase between these two cases of approximately 2.35×10^{-6} per reactor year. The risk increase for both analyses was dominated by the increased likelihood of AFW pump failure following the Loss of ESW initiating event. The overall CDF difference between these two cases using the CNP full-scope PRA model is approximately 3.1×10^{-6} per reactor year. The major cause of the difference between these two cases is dominated (about 90 percent) by random and common cause failure of the MDAFP room coolers following a transient scenario. The contribution to overall risk due to eliminating the impact of HELB events on AFW operability is described in response to Question 3F and significantly offsets these small increases in risk described in response to this question.

A bounding, worst-case estimate of the effect of the proposed modification on the seismic CNP PRA would be to assume that the new AFW dependence on ESW could cause seismically induced loss of ESW events to proceed to core damage without the possibility of mitigation. Given this assumption, the increase in seismic CDF due to the modifications is on the order of 2×10^{-7} per reactor year.

Random failures of the ESW system were not significant contributors to the CDF determined by the CNP fire risk analysis. Consequently, the predominant impact of the new AFW dependence on ESW would be seen in sequences in which a fire causes a loss of ESW as the initiating event. However, since the CNP fire risk analysis was conducted from a bounding perspective, a detailed consideration of the new AFW dependence on ESW cannot result in any increase in the results of the CNP fire risk analysis.

NRC Question 3F

"Provide the following information associated with the risk assessment:

F. Additional discussion on the safety benefits of the modifications is important; in particular, the risk decrease from the elimination of high-energy line break scenario vulnerability should be discussed."

I&M Response to Question 3F

The risk assessment performed to evaluate the existing risk impact due to the failure of the current design to protect the AFW pumps from the impact of HELB events required revision of the CNP full-scope PRA model. This revision was required since the current CNP full-scope PRA model assumes that HELB events do not impact AFW pump operation. The CNP full-scope PRA model was revised to include HELB events outside containment that could adversely affect the operation of the AFW pumps. As described in the answer to Question 3B, the overall HELB IEF value conservatively used in this risk analysis is 1×10^{-2} per reactor year. The IEF for two sub-categories of HELB events was determined based on piping ratios obtained from walkdowns of the turbine building and auxiliary building HELB piping. The first HELB sub-category assumes that all three AFW pumps fail immediately for breaks in all feedwater and main steam system piping with a pipe diameter greater than or equal to 8 inches that is located at the same elevation as the AFW pump rooms. In the second HELB sub-category, other turbine building HELB piping is assumed to have a 10 percent probability of failing all three AFW pumps. HELB breaks in the auxiliary building are assumed not to fail the AFW pumps.

Comparing the first HELB case with the existing CNP PRA model suggests that the HELB contribution to risk could be significant. This case, when compared with the existing CNP full-scope PRA model, suggests that the total CDF increase due to including HELB scenarios to the existing CNP full-scope PRA model is 3.70×10^{-5} per reactor year. This risk increase is due primarily to increasing the overall HELB IEF assumed in the revised CNP full-scope PRA model (from 3.3×10^{-4} per reactor year to 1.0×10^{-2} per reactor year) and adding two new single-failure core damage scenarios (a failure of feed and bleed, and a failure to initiate emergency core cooling system recirculation) following either of the two new HELB sub-categories.

A second case that credits the HELB protection features of the proposed modification by assuming that any HELB located outside containment has a 1 percent chance of failing all three AFW pumps results in a total CDF increase of 5.20×10^{-6} per reactor year.

Based on this risk analysis approach, the net impact on overall CDF due to incorporating the HELB protection features of the modifications is a decrease of approximately 3.2×10^{-5} per reactor year.

NRC Question 3G

"Provide the following information associated with the risk assessment:

G. One way to evaluate the risk is to use the Regulatory Guide (RG) 1.174 criteria. Does the risk impact of the proposed change meet the intent of the acceptance criteria in RG 1.174 in terms of core damage frequency and large early release frequency?"

I&M Response to Question 3G

The results of the risk analyses performed using simplified event trees and the CNP full-scope PRA model are shown in the table below.

Table PRA-1
CDF Comparisons Between the Simplified Event Tree and CNP Full-Scope PRA Model Analyses

Case #	Case Description	Model Description		
		Simplified Loss of ESW Event Tree CDF	**Full PRA Loss of ESW Event Tree CDF**	**Full PRA Model CDF**
1	Base Case	5.19×10^{-6}	5.48×10^{-6}	7.06×10^{-5}
2	Case 1 + TDAFP survives w/o room cooling	7.20×10^{-6}	7.83×10^{-6}	7.37×10^{-5}
3	Case 2 + Loss of ESW Procedure Credited	3.29×10^{-6}	3.73×10^{-6}	6.91×10^{-5}
4	Case 3 + MDAFP survives w/o room cooling	1.02×10^{-6}	1.11×10^{-6}	6.41×10^{-5}
		Simplified Steam Line Break Event Tree CDF	**Full PRA Steam Line Break Event Tree CDF**	**Full PRA Model CDF**
5	Base Case	6.32×10^{-7}	7.39×10^{-7}	7.06×10^{-5}
6	Case 5 + Large Turbine Building HELB Event	3.47×10^{-5}	3.77×10^{-5}	1.08×10^{-4}
7	Case 5 + Other HELB Events in Turbine/Auxiliary Building	5.09×10^{-6}	5.94×10^{-6}	7.58×10^{-5}
		Combined Simplified Event Tree Model Change in CDF	**Full PRA Model Change in CDF**	
8	(Case 4 –Case 1) + (Case 7 - Case 6)	-3.38×10^{-5}	-3.9×10^{-5}	Final Configuration versus Worst-Case Configuration
9	(Case 4 –Case 1) + (Case 7 - Case 5)	$+2.88 \times 10^{-7}$	-1.3×10^{-6}	Final Configuration versus Base Case Configuration

Case 8 in Table PRA-1 represents the overall risk decrease provided by the proposed modification and procedure enhancements when compared to the existing plant configuration. This decrease in risk is the difference between the existing configuration that does not adequately protect the AFW pumps from HELB events and the final configuration that protects the AFW pumps and ensures proper operator response to a loss of room cooling and loss of ESW event.

A similar analysis was performed to characterize the effects on large early release frequency (LERF) behavior associated with implementing the proposed modification and procedure enhancements, HELB scenarios, and the combination of these effects. For the risk assessment of LERF, only analyses using the CNP full-scope PRA were performed for the same cases as the CDF risk assessment. Based on the CNP full-scope PRA model, implementation of the proposed modification decreases overall LERF.

Since the risk analyses performed demonstrate a reduction in overall risk, the intent of the acceptance criteria in Regulatory Guide 1.174 for CDF and LERF are met by the proposed modification.